

NON-PUBLIC?: N  
ACCESSION #: 9106110274  
LICENSEE EVENT REPORT (LER)

FACILITY NAME: HOPE CREEK GENERATING STATION PAGE: 1 OF 6

DOCKET NUMBER: 05000354

TITLE: REACTOR SCRAM ON LOW WATER LEVEL DUE TO PERSONNEL  
ERROR

EVENT DATE: 05/07/91 LER #: 91-008-00 REPORT DATE: 06/06/91

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR  
SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: Robin Ritzman, Lead Engineer - TELEPHONE: (609) 339-3431  
Technical

COMPONENT FAILURE DESCRIPTION:

CAUSE: SYSTEM: COMPONENT: MANUFACTURER:

REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On 5/7/91, at 2102, a Reactor Scram occurred due to a low Reactor Vessel water level. Conditions leading to the Scram were the 'A' Channel Feedwater Level Control indication alarming with a high level, causing the feedpumps to back down. This was followed by the 'C' Channel alarming at 30" and a subsequent Reactor Scram. All control rods were verified to be inserted and plant systems responded as expected, with minor exceptions as noted in the text of this report. Follow-up investigation determined that the most probable cause of the event was personnel error when a Controls Technician inadvertently connected a current source to the wrong transmitter while performing a surveillance test. Although the most probable root cause was personnel error, a less than adequate cabinet design also contributed to the event. Immediate corrective actions included counseling the Controls Technician with respect to self-verification and att

ntion to detail and implementing a design change to install test switches with input jacks and to label the input jacks with a channel designator. The surveillance procedure was revised in accordance with the design change and is being reviewed for additional enhancements.

END OF ABSTRACT

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#### PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor (BWR/4)  
Reactor Protection (EIS Designation: JC)  
Reactor Recirculation (EIS Designation: AD)  
High Pressure Coolant Injection (EIS Designation: BJ)  
Reactor Core Isolation Cooling (EIS Designation: BN)  
Primary Containment Isolation (EIS Designation: JM)  
Nuclear Instrumentation (EIS Designation: IG)  
Feedwater (EIS Designation: SJ)  
Control Room Ventilation (EIS Designation: VI)  
Service Water (EIS Designation: BI)

#### IDENTIFICATION OF OCCURRENCE

Reactor Scram on Low Reactor Vessel Water Level Due to Personnel Error

Event Date: 5/7/91

Event Time: 2102

This LER was initiated by Incident Report No. 91-068

#### CONDITIONS PRIOR TO OCCURRENCE

Plant in OPERATIONAL CONDITION 1 (Power Operation), Reactor power 100%, unit load 1110 Mwe.

#### DESCRIPTION OF OCCURRENCE

On May 7, 1991, at 2102, with the Unit operating at 100% Reactor Power (1110 MWe), and all major equipment except Channel 'A' Control Room Ventilation and the 'B' Station Service Water Pump in service, a Reactor Scram occurred due to a low reactor vessel water level. Conditions leading to the Scram were the 'A' Channel Feedwater Level Control level indication alarming with a high level of 60.75 inches, causing the feedpumps to back down. Eleven seconds later, Channel 'C' alarmed at Level 4 (30 inches), 8 seconds later a Level 3 Reactor Scram occurred.

The Reactor Protection System operated normally to shutdown the Reactor. A recirculation pump runback normally occurs at Level 3, however due to the Channel 'A' high level indication, this did not occur. The High Pressure Coolant Injection (HPCI) and the Reactor Core Isolation Cooling (RCIC) Systems initiated at Level 2 (-38") as per design, and were terminated by the operator. Neither HPCI nor RCIC injected to the vessel.

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#### DESCRIPTION OF OCCURRENCE, CONT'D

When the -38" Reactor water level signal was received, the Primary Containment Isolation System (PCIS) and the Nuclear Steam Supply Shutoff System (NSSSS) isolated and the Redundant Reactivity Control System (RRCS) actuated. No discrepancies were noted except as listed below:

1. The 'B' Automatic Rod Insertion (ARI) valve open light stayed on after resetting ARI and RRCS.
2. The Nuclear Instrumentation (NI) drive-in button failed to drive in automatically, and needed to be held in to insert the NI's.
3. The 'A' Circulating Water Pump Discharge Valve Open Indication failed to go out when the pump was removed from service.
4. The #3, 4, and 5 Feedwater Heater Bypass Valve did not lose its closed indication upon opening.

After plant conditions stabilized, a four hour non-emergency report was initiated in accordance with 10CFR50.72 due to the automatic initiation of an Engineered Safety Feature.

The above noted discrepancies were corrected prior to restart.

#### ANALYSIS OF OCCURRENCE

At the time of the event, surveillance testing was being conducted on the Reactor Feed Pump Turbine Level 8 (high level) Channel 'B' trip instrumentation, which also provides level information to the Feedwater Level Control System.

At 2037, the Channel 'B' surveillance was signed on by the operating shift and the 'A' Channel was selected for input to the level control. The Controls Technicians proceeded with the 'B' channel surveillance.

The surveillance requires the 'B' transmitter to be disconnected from the instrument loop, a current source to be connected to the loop, and an increasing level to be simulated until the trip setpoint is reached. At 2056, the 'B' channel level failed downscale (indicating that the 'B' transmitter was disconnected). The 'C' level indication was reading 35", as was wide range level indication.

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#### ANALYSIS OF OCCURRENCE, CONT'D

At 2102:39, the 'A' Channel level indication alarmed with a high level of 60.75 inches, causing the feedpumps to back down. At 2102:50, Channel 'C' alarmed at Level 4 (30 inches), followed 8 seconds later (2102:58) with a Level 3 Reactor Scram.

At 2104:05, Channel 'A' level indication returned to normal, followed at 2104:13 with level 'B' indication jumping from downscale to 52". During the remaining time of the event, Channel 'A' level indicated properly and Channel 'B' remained at its high level.

As part of the troubleshooting plan, System Engineering witnessed the performance of the surveillance as written to determine if equipment failure or procedural inadequacy may have contributed to the event. The procedure was technically accurate as written; however, some human factors concerns were identified. Several components were also tested, with no failures noted.

While studying design documentation in an attempt to determine a possible cause of the Scram, a potential personnel error was postulated. It was postulated that the technician may have correctly lifted the leads, but then incorrectly installed the current source.

Lifting the leads to the 'B' transmitter would account for the fact that 'B' indicated downscale in the Control Room. Connecting the current source to the 'A' transmitter would account for the response of the Channel 'A' transmitter. This postulated error was recreated. The results showed a similarity between the two GETARS traces.

Due to this similarity, along with the inability to recreate the problem by following the procedure or through component testing, it was concluded that the technician inadvertently connected the current source to the wrong transmitter.

#### APPARENT CAUSE OF OCCURRENCE

The initiating event of this Scram was low level in the Reactor. This was caused by a high level indication in the 'A' channel of the Feedwater Level Control System. The high level indication caused the feedpumps to back down, resulting in the Scram on Low Reactor Level.

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#### APPARENT CAUSE OF OCCURRENCE, CONT'D

As previously discussed, the most probable cause of the Scram was personnel error when the technician inadvertently connected the current source to the wrong transmitter. Also, a less than adequate cabinet design contributed to the event. The deficiencies in the cabinet design have been corrected.

#### PREVIOUS OCCURRENCES

There have been two previous reportable events that were caused in part by incorrectly installed test equipment. In both instances, testability was a contributing factor.

The first (Ref: LER 87-017) occurred on 2/24/87 at 1022, when a Reactor Scram occurred during the performance of a surveillance on the 'C' Reactor Vessel Level 8 Trip. The direct cause of this event was determined to be personnel error in placing ohm meter leads on the wrong contacts. Testability was also considered to be a contributing environmental factor.

The other (Ref: LER 90-022) occurred on 10/22/90 at 1405, when the 'A' Core Spray Pump inadvertently started. The primary cause of this event was a personnel error on the part of a Controls Technician who was performing a surveillance test. Testability concerns in the relay cabinet in which the test was being conducted contributed to the personnel error; however, the procedurally required sequence of independent verification of test switch lead installation also contributed to the event. The surveillance procedure was reviewed for adequacy of independent verification. It was determined that the steps for verification of landing the test switch leads could be enhanced to include a second verification after the leads are landed.

#### SAFETY SIGNIFICANCE

The Feedwater Control System controls the flow of feedwater into the Reactor Vessel to maintain the water level within the proper limits during normal plant operations. Normal Reactor level is approximately +35". Subsequent to this Scram, level was immediately restored using

feedwater. If feedwater had been unavailable during this event, other sources of water were available. For example, HPCI and RCIC initiated and were available for injection.

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#### SAFETY SIGNIFICANCE, CONT'D

Therefore, the safety significance of this event was minimal, as a sufficient inventory of water was available and a plant scram is an analyzed transient. Additionally, all systems (with previously noted minor exceptions), responded as expected. This event posed no threat to the health and safety of the general public.

#### CORRECTIVE ACTIONS

1. A design change has been implemented to install test switches with input jacks and to label the input jacks with a channel designator in addition to the current identifiers.
2. This surveillance procedure has been revised in accordance with the design change and is being reviewed for additional enhancements.
3. The Controls Technician has been counseled with respect to self-verification and attention to detail. A concerted effort to heighten the awareness of personnel performance has been developed based on the guidelines of INPO Good Practice 90-001, "Increasing Personnel Awareness of Frequent Causes of Human Performance Problems." This effort will be given to all plant personnel via a rolldown plan, beginning with the General Manager of the plant to his direct reports.

Sincerely,

J.J. Hagan  
General Manager -  
Hope Creek Operations

RAR/

SORC Mtg. 91-055

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PSE&G

Public Service Electric and Gas Company  
P.O. Box 236 Hancocks Bridge, New Jersey 08038

Hope Creek Operations

June 6, 1991

U. S. Nuclear Regulatory Commission  
Document Control Desk  
Washington, DC 20555

Dear Sir:

HOPE CREEK GENERATING STATION  
DOCKET NO. 50-354  
UNIT NO. 1  
LICENSEE EVENT REPORT 91-008-00

This Licensee Event Report is being submitted pursuant to the requirements of 10CFR50.73(a)(2)(iv).

Sincerely,

J.J. Hagan  
General Manager -  
Hope Creek Operations

RAR/

Attachment  
SORC Mtg. 91-055

C Distribution

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